NON-PUBLIC?: N
ACCESSION #: 9505240276
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Arkansas Nuclear One - Unit 1 PAGE: 1 OF 5

DOCKET NUMBER: 05000313

TITLE: Reactor Trip Initiated by Main Turbine Generator
Protective Circuitry as a Result of a Logic Circuit
Ground Caused by Vibration Induced Insulation Wear
EVENT DATE: 04/20/95 LER #: 95-005-00 REPORT DATE: 05/19/95

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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Safety and Licensing Specialist

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On April 20, 1995, at approximately 0313, an automatic reactor trip was initiated by the Reactor Protection System as a result of a main turbine trip. The turbine trip was initiated by a generator lockout which was caused by a negative sequence relay (NSR) actuation. All control rods inserted into the core, as designed, and immediate operator actions were accomplished with no complications. The plant was safely taken to Hot Shutdown although some minor abnormalities occurred post-trip. Investigation into the cause of the trip identified a ground in the NSR circuitry which caused a current imbalance to the NSR which resulted in actuation of the relay. The most probable cause of the ground was vibration induced wear of wiring insulation inside an electrical junction box. The grounded wire was repaired and a rubber insulation mat was installed in the junction box to prevent vibration induced insulation

wear. other similar junction boxes were examined and no additional discrepancies were identified

END OF ABSTRACT

A. Plant Status

At the time of this event, Arkansas Nuclear One, Unit One (ANO-1) was operating at approximately 100 percent power. Reactor Coolant System (RCS) AB! temperature was 579 degrees and RCS pressure was 2155 psig.

B. Event Description

On April 20, 1995, at approximately 0313, an automatic reactor trip was initiated by the Reactor Protection System (RPS)JE! as a result of a main turbine trip. The turbine trip was initiated by a generator lockout which was caused by a negative sequence relay (NSR) actuation. All control rods inserted into the core, as designed, and immediate post trip operator actions were accomplished with no significant complications.

The NSR is intended to protect the main generator from thermal damage due to negative sequence current caused by system faults or an open phase condition. The NSR is set to coordinate with system protective relays and will operate to lockout the main generator if a fault or open phase condition occurs. A generator lockout initiates a turbine trip which will, in turn, initiate a RPS trip of the reactor if power is above 43 percent.

During the post trip response, the Main Steam Safety Valves (MSSVs) opened as expected. However, one valve (PSV-2684) appeared to remain open longer than normal. Operators initiated action to reduce the "B" Once Through Steam Generator (OTSG) pressure to assist the MSSV in closing. The "B" Turbine Bypass Valve (TBV) was rapidly opened to approximately 50 percent, at which time PSV-2684 seated. The "B" TBV was then returned to automatic control and the "B" OTSG pressure began to increase slowly resulting in PSV-2684 opening again. The TBV was again placed in manual and OTSG pressure was lowered until the MSSV seated. The TBV was left in manual control until the plant was stabilized at Hot Shutdown.

Several abnormal system responses were observed after the trip:

o Following the reactor trip, both main feedwater pumps (MFPs) ran back to minimum speed, as designed. Upon reaching appropriate OTSG levels, the s should be automatically released from Rapid Feedwater Reduction (RFR) to maintain OTSG levels. Operators observed that the "B" transferred to manual instead of returning to automatic control, as required. The "B" Hand/Auto (H/A) station signal was matched to the "A" MFP H/A station signal, and returned to automatic. This condition did not present a significant challenge to the operators and there were no further problems with control.

o Over a period of approximately one hour following the trip, condenser vacuum degraded to approximately 20.4 inches Hg. Condenser vacuum pump C5B was found to be running in the "Holding Mode" while pump C5A was running in the "Hogging Mode". At the time of the trip, C5A was the running pump and should have been capable of maintaining condenser vacuum in the absence of significant air in-leakage. However, C5B should have automatically shifted to the "Hogging Mode" when vacuum decreased to 24 inches Hg. The operators manually shifted C5B to the "Hogging Mode" and increased Moisture Separator Reheater (MSR) seal pressure to approximately 7 psig. Condenser vacuum then returned to normal.

o Approximately one hour after the trip, channel "A" of the Emergency Feedwater Initiation and Control (EFIC) system received a half trip as a result of the failure of a +5 VDC power supply. This condition resulted in the loss of Train "A" OTSG level indication, a low level initiate to Train "A" EFIC, and the loss of control function for Emergency Feedwater control valves CV-2646 and CV-2648 and remote control of Atmospheric Dump Valve (ADV) CV-2668. As a contingency in the event that the ADV might be needed, operators opened the valve locally. The ADV block valves remained closed and no steam release occurred through the ADVs. No EFW actuation occurred as a result of the power supply failure.

C. Root Cause

Investigation into the cause of the negative sequence relay trip identified a ground on the "B" phase current transformer (CT) lead from the transformer to the relay. Further investigation revealed brittle and cracked insulation on the "B" phase wires inside a junction box at the generator. Evidence of arcing to ground was found at that location. Indications of wear resulting from the CT wiring rubbing against the cover plate was identified on one of the

CT leads. No other brittle or cracked wiring was identified.

The most probable root cause of this event was determined to be vibration induced wear of the CT wiring that resulted in a ground, causing a current imbalance to the negative sequence relay which resulted in relay actuation and ultimately, the reactor trip.

PSV 2684 was lift pressure tested as a conservative measure to determine the potential blowdown range of the valve based on actual setpoint since it was the last valve to reseat and was the cause of the perception by the operators that a valve was open too long. The as-found setpoint was 1037 psig. This would correlate to an acceptable blowdown range of 943 to 1006 psig. A review of all the Safety Parameter Display System data showed that PSV-2684 responded normally on blowdown and reseat through several valve strokes as the valve reseated within the acceptable blowdown range.

The cause of the failure of the "B" to shift back to automatic control was determined to be foreign material on a module connector which provides power to a RFR logic relay coil. A 1/2 inch diameter

calibration sticker was found covering a module connector which prevented proper connection to the cabinet back plane.

The probable cause for the degraded condenser vacuum over a one hour period was significant condenser air in-leakage. A primary indicator of this leakage was a corresponding increase in vacuum when steam seal pressure was increased on the MSR relief valves. It was noted that MSR pressure decreased in parallel with the condenser vacuum drop during the evolution. Vacuum loss cannot be attributed to operation of only one pump in the hogging mode The pump performance curve indicates one pump has the ability to remove nearly 800 CFM of non-condensables. This suggests one hogging pump was not able to keep up with the "air in-leakage" rate. It appears the "A" pump switched to hogging mode at the 23" Hg setpoint while the "B" pump remained in holding mode for nearly 90 minutes. Subsequent trouble-shooting determined that the setpoint for the pressure switch which controls the "B" pump was three inches low (21" Hg versus 24" Hg).

The failure of the +5 VDC Power Supply was apparently due to a failure of the voltage regulating circuit within the supply which was unrelated to the reactor trip. The loss of the ADV remote control is also directly related to the loss of the +5 VDC power supply. This supply provides integrated circuit logic power to the compensation module and control module portions of the EFIC Channel "A" ADV control circuit. The compensation modules provide density compensation for the EFIC OTSG Level inputs based on OTSG pressure. The control module provides level and pressure control for the OTSGs by modulating the Emergency Feedwater (EFW) flow control valves (CV-2646 and CV-2648) and the ADV (CV-2668). The failure of the power supply resulted in the loss of both EFW Flow Control Valves and CV-2668 control in either automatic or manual.

D. Corrective Actions
Immediate:
o The grounded wire from the "B" phase CT was repaired and a rubber insulation mat installed in the junction box to prevent vibration induced insulation wear.

o An inspection and megger check of the leads from the CTs of all three phases was performed. No additional wear or unsatisfactory megger readings were identified.

o PSV-2684 was lift pressure tested and verified to be operable.

o The foreign material was removed from the MFP control contacts and the circuit was proven operable.

o Condenser vacuum pump C-5B pressure switch was reset to 25" Hg increasing. In addition, MSR relief valve seal steam pressure has been increased and set in accordance with plant operating procedures.

E. Safety Significance
The turbine and reactor protective circuitry performed as designed during this event and the plant was safely taken to Hot Shutdown conditions. The operators expeditiously and properly compensated for all identified post trip abnormalities. Therefore, this event is considered to be of minimal safety significance.

F. Basis for Reportability A reactor trip is a reportable event in accordance with 10CFR50.73(a)(2)(iv). This event was also reported to the NRC Operations Center at 0453 CST on April 20, 1995, pursuant to 10CFR50.72(b)(2)(ii).

This event was also reported to the NRC Operations Center in accordance with 10CFR50.72 at 0453 CST on April 20, 1995.

G. Additional Information
LER 50-313/93-001-00 reported a reactor trip which resulted from two grounds on the 125 VDC system. One of the grounds was caused by vibration induced wear of wiring that passed through an ungrommeted hole in the wall of the main turbine front standard. The corrective actions associated with this LER were focused on the main turbine front standard, MFP control circuitry, and wiring passing through panel walls and could not reasonably be expected to identify the condition that caused the trip discussed in this report.

Energy Industry Identification System (EIIS) codes are identified in the text as XX!.

ATTACHMENT TO 9505240276 PAGE 1 OF 2

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May 19, 1995

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U. S. Nuclear Regulatory Commission
Document Control Desk
Mail Station P1-137
Washington, DC 20555

Subject: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
Licensee Event Report 50-313/95-005-00

Gentlemen:

In accordance with 10CFR50.73(a)(2)(iv), enclosed is the subject report concerning a reactor trip.

Very truly yours,

Dwight C. Mims
Director, Licensing

DCM/rhs

enclosure

ATTACHMENT TO 9505240276 PAGE 2 OF 2

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